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June 16, 1995

U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1; Docket No. 50-317; License No. DPR 53
Licensee Event Report 94-007, Revision 01
Reactor Trip Caused by Closure of Turbine Stop Valves

The attached Licensee Event Report Supplement is being sent to you to fulfill our commitment in the original report. Should you have questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

CHC/CDS/bjd

Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
L. B. Marsh, NRC
D. G. McDonald, Jr., NRC
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NRC FORM 366 (5-92) U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Calvert Cliffs, Unit 1X	05000 317	1 OF 12

TITLE (4)
Reactor Trip Caused by Closure of Turbine Stop Valves

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
07	19	94	94	-- 007 --	01	06	16	95		05000
										05000

OPERATING MODE (9): 1
POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR :(Check one or more) (11)

	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include Area Code)
Craig D. Sly, Compliance Engineer	410-260-4858

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	T G	C B D	G080	N					

SUPPLEMENTAL REPORT EXPECTED (14)

				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-space typewritten lines) (16)

On July 19, 1994 at 1824 Calvert Cliffs Unit 1 tripped from 100 percent power due to a Reactor Protection System (RPS) actuation. The RPS actuation was the result of low steam generator water levels due to level shrink after all four main turbine stop valves (MTSV) unexpectedly closed. During the resulting transient both Reactor Coolant System (RCS) power-operated relief valves opened and one code safety relief valve (RV) opened, closed, and then began leaking by its seat at approximately 25 gpm.

The event did not result in any significant potential or actual nuclear or personnel safety consequences.

Short-term corrective actions to support a safe Unit restart were completed. The RV manufacturer has initiated process improvements to prevent recurrence of inadequate staking of a disc holder to bellows assembly inside the RV. The root cause of the MTSV closure has been incorporated into an ongoing turbine EHC Systems improvement effort.

I. DESCRIPTION OF EVENT

On July 19, 1994 at 1824, the Calvert Cliffs Unit 1 reactor tripped from 100 percent power due to a valid automatic actuation of the Reactor Protection System (RPS). The RPS signal was the result of low Steam Generator (SG) water levels due to level shrink after all four main turbine stop valves (MTSVs) unexpectedly closed. This RPS system response was as designed. During the resulting transient, both Reactor Coolant System (RCS) power-operated relief valves (PORVs) (ERV-402 and 404) opened. In addition, one code safety relief valve (RV-201) opened, closed, and then began leaking by its seat at approximately 25 gpm.

During shift turnover on July 19, 1994, Operators in the Control Room received multiple alarms on several Control Room panels. Shortly thereafter, a reactor trip occurred. Operators initiated Emergency Operating Procedure (EOP)-0, "Post Trip Immediate Actions." An immediate scan of the Control Room panels found that Quench Tank parameters were in alarm and all four acoustic monitor indicators for the two RVs and two ERVs indicated flow. EOP-0 Reactivity Control and Vital Auxiliary Safety Functions were verified to be satisfactory and operators started Pressure and Inventory Control. A subsequent scan of the acoustic monitors found only ERV-402 and RV-201 indicating flow via their acoustic monitor.

After the trip, RCS pressure promptly spiked up to 2410 psia and then dropped rapidly to about 1770 psia before pressurizer level and pressure were stabilized and restored. Pressurizer pressure did not reach the safety injection actuation signal setpoint of less than 1725 psia. Steam Generator levels decreased to about -112 inches within one minute, then stabilized and started to slowly trend downward.

About two minutes after the trip, the Quench Tank pressure suddenly dropped to a very low pressure indicating its rupture disc had ruptured. About eight minutes after the trip, with water levels in SG 11 at -155 inches, and SG 12 at -160 inches, Control Room Operators manually started 13 Auxiliary Feedwater (AFW) Pump in anticipation of an impending Auxiliary Feedwater Actuation Signal (AFAS) at -170 inches. The AFW Pump restored positive-level trends in both SGs.

EOP-0 was exited and EOP-1, "Reactor Trip," was entered. The acoustic monitor for RV-201 still indicated flow. Leakage from RV-201 was estimated to be about 25 gpm indicating that RV-201 was not fully open, but was leaking. At 1842, the ERV block valves were shut to confirm that the leakage was not from ERVs 402 or 404. RV-201 continued to show signs of leakage. By 2045, the plant had stabilized from the trip and post-trip actions were completed. Cooldown of the plant was commenced at 2130, with some hold points in order to facilitate troubleshooting and analysis of the ERVs, RVs, and block valves. The cooldown was conducted in accordance with Operating Procedure (OP)-5, "Plant Shutdown from Hot Standby to Cold Shutdown."

The Post-Trip Review concluded that the reactor tripped on low SG level and that a trip on RCS high pressure was less than one second away. The analysis also found that SG pressures were rising and turbine steam chest pressure was rapidly dropping just prior to the trip. Since the main steam isolation valves did not close, it was concluded that closure of the MTSVs had initiated the transient.

II. CAUSE OF EVENT

A. CAUSE OF REACTOR TRIP

A detailed analysis of the reactor trip Sequence of Event Printout revealed that turbine steam chest pressure and SG levels were rapidly decreasing just prior to the trip. Reactor Coolant System pressure and SG pressures were rapidly increasing at the same time. The closure of the MTSVs caused SG pressures to promptly increase and level to shrink. Correspondingly, the ability of the SGs to remove heat from the RCS was reduced and RCS pressure and cold leg temperature promptly increased. The reactor tripped due to low SG level (-50 inches). In the first second following the trip, high pressurizer pressure reactor trip signals were received (2385 psia). Also within one second, acoustic indications were received from sensors downstream of ERVs 402 and 404, and primary RVs 200 and 201. Due to the close proximity of the piping and relief valves, it is difficult to positively discern which valves are open and which are closed based on acoustic monitor indication alone.

Troubleshooting of the MTSV electro-hydraulic control (EHC) system found that an intermittent failure of a Servo Amplifier Demodulator Indicator (SADI) board in Turbine EHC Cabinet 1T11 was the most probable cause of the MTSVs closing. This board supplies a signal to a hydraulic Servo valve controlling the No. 2 MTSV. MTSV-2 is the master valve for the other MTSVs. Failure of the SADI board caused MTSV-2 to close. As MTSV-2 closed, it caused the other three MTSVs to receive close signals. The result was the simultaneous closure of all four MTSVs.

The SADI board was sent to the turbine vendor for a detailed failure analysis. The test consisted of placing the SADI board in an oven with a known electrical input and monitoring the output. Over a one month period, the oven temperature was gradually raised to 125 degrees Fahrenheit. No abnormal output was noted during the test. Therefore, the results of the vendor analysis were inconclusive. Testing was unable to reproduce the SADI board intermittent failure noted at the plant during the troubleshooting efforts after the trip. The SADI board was subsequently sent to an independent consultant specializing in the identification of hardware failure modes.

The consultant concluded that the most probable cause of the SADI board failure was a failed transistor (Q3) on the board (see Figure 2). Failure of the transistor was intermittent and most likely due to age related degradation. Destructive examination found a small conductive particle loose inside the transistor. This loose particle is considered the most likely cause of the failed transistor. The electromagnetic field created during energization of the transistor probably attracted the loose particle to the base-collector junction causing a short circuit and signal failure similar to the one observed during troubleshooting after the trip. Computer modeling and actual shorting of the collector to the base were utilized to verify the behavior of the circuit. The behavior of the circuit with a collector-base short circuit was verified to be very similar to the observed behavior of the SADI board during troubleshooting.

Examination of other transistors on the SADI board revealed signs of manufacturing defects and/or age related degradation. The consultant also observed that the SADI board was in good overall condition and very well constructed with the highest quality components available at the time of production. The SADI boards components, with the exception of the problems noted with the transistors, were not considered susceptible to similar wearout mechanisms. Both the turbine vendor and an independent consultant who is performing an analysis of our Turbine EHC Systems to identify improvements have stated that the large population of similar boards in service at fossil and nuclear facilities have been very reliable.

B. CAUSE OF RELIEF VALVE LEAKAGE

As stated earlier, initial plant indications were that both PORVs lifted less than one second after the trip. The maximum recorded pressure of the RCS after the trip was 2410 psia. The setpoints of the relief valves are as follows:

Code Safety RV-200	2500 +/- 1 percent psia
Code Safety RV-201	2565 +/- 1 percent psia
PORVs ERV-402 and 404	2385 +/- 15 psia

After the trip, it was verified that the ERV-402 and ERV-404 did open, as designed, when RCS pressure reached their setpoints. The valves were verified electrically energized (open) for about 1 second. Maximum RCS pressure reached was 2410 psia and quickly decreased after ERV-402 and 404 opened. Six seconds after the trip the acoustic monitors for both ERVs and RV-200 indicated these valves were closed. The acoustic monitor to RV-201 indicated it was leaking. Reactor Coolant System pressure continued to decrease to approximately 1770 psia. Pressurizer level was restored and pressure subsequently stabilized at 1900-1940 psia. RV-201 acoustic monitor continued to indicate leakage. These plant parameters combined with acoustic monitor indications, valve outlet pipe temperatures, and RCS leak rate determinations led operators and engineers to estimate that RV-201 was leaking at about 25 gpm (less than the capacity of one charging pump).

We have performed a detailed root cause analysis for RV-201. This RCA found that RV-201 had indeed lifted, partially reseated, and began leaking by its seat after the event. The root cause analysis consisted of the following:

1. Onsite physical examination and cold lift testing using nitrogen.

2. The valve was sent to Wylie Laboratories for a lift test using steam and disassembly to examine internal parts.

3. Some damaged parts were sent to Dresser for close-up visual inspection and analysis.

Based on the hardware root cause analyses conducted on RV-201, the cause of the premature lift was determined to be an improperly staked disc holder (see Figure 1). This disc holder was improperly staked to the bellows assembly, allowing the disc to rotate and drift downward toward the lower adjusting ring. This valve specific deficiency effectively lowered its setpoint when subjected to other contributing factors such as valve leakage, elevated RCS pressure, and flow-related vibration created by opening of 1-ERV-404.

The cause of the subsequent RV-201 leakage was that the valve failed to properly reseat after the lift transient due to damage and misalignment of the valve's internal components resulting from the valve lift and/or flutter occurring during the transient. Valve "flutter" is a rapid reciprocating motion of its internal parts in which its disc does not contact the seat.

C. OTHER CONCERNS

During the course of the event investigation, several other issues were identified, including the following:

1. The computer post-trip data was not available after the event. Although some Sequence of Event Data did printout and was available, some data that normally becomes available did not printout. The problem was found to be a computer software deficiency.

2. The main feedwater system did not deliver enough flow to the SGs to restore positive level trends. As a result, operators decided to start 13 AFW pump about eight minutes after the trip to restore levels and avoid an AFAS. An analysis was already underway to determine if an AFAS actuation should be expected after a reactor trip, with main feedwater flow available.

3. The effects of the 5000 gallons of reactor coolant discharged to the containment floor and general area in the vicinity of the Quench Tank and the effect of this moisture on equipment inside containment were evaluated. They were determined to have no effect on the operability of equipment in Containment.

4. ERV-404 had a leaking pilot valve. This conclusion was based upon a significant increase in valve pilot temperature after its block valve MOV-405 was opened while the RCS was at 1000 psia.

5. Some fans inside the EHC cabinet, which contain the SADI board, had previously failed.

III. ANALYSIS OF EVENT

A. ACTUAL SAFETY CONSEQUENCES

This event resulted in no actual safety consequences.

1. There were no personnel injuries or personnel exposures as a result of the event. The event did not place any personnel in danger of being injured or exposed to radiation.

2. There were no significant radiological releases or off-site doses as a result of the event. Two main steam system atmospheric dump valves and possibly up to two Main Steam Safety Valves (MSSVs) lifted as a result of the secondary system pressure transient.

3. There was no damage to plant equipment except the Quench Tank rupture disk, which performed its design function.

4. The vendor has evaluated the as-found condition of the RV-201 internals and has concluded that the valve would have been able to lift and perform its safety function.

B. POTENTIAL SAFETY CONSEQUENCES

Although this event did represent a challenge to both a loss-of-coolant accident (LOCA) and the RCS pressure upset limit, the potential consequences were mitigated by:

1. This event constituted a loss of load event as analyzed in Chapter 14 of our Updated Final Safety Analysis Report (UFSAR). The transient was fully bounded by that analysis. The Chapter 14 analysis credits the High Pressurizer Pressure Trip, actuation of two primary code safety valves, and opening of two MSSVs to limit the transient. All of that equipment was available and satisfactorily performed its safety function where challenged.

   In addition to that protection credited by the UFSAR Safety Analysis, the actual event was mitigated by the fact that a low steam generator level trip preceded the high pressure actuation by about one second; the PORVs responded as designed to limit the RCS pressure transient; moderator temperature coefficient was less limiting than assumed in the analysis; and the Main Steam System Atmospheric Dump Valves functioned to supplement the MSSVs in removing steam generator heat. Both RVs were available to limit RCS pressure to a value less than the RCS pressure upset limit of 2750 psia. Post trip investigation of RV-200 verified that the valve was fully operable and capable of performing its safety function. The root cause analysis on RV-201 confirmed our expectation that the valve would have performed its intended safety function.

2. This event is closely modeled by the RCS Depressurization Accident Analysis in our UFSAR. An RCS depressurization event is defined as a rapid, uncontrolled decrease in the RCS, other than a LOCA. Inadvertent opening of one RV or both PORVs during steady-state operation would result in such an event. The most limiting case analyzed in the UFSAR is the inadvertent opening of both PORVs since two PORVs have a larger relieving capacity than one RV. This Safety Analysis fully bounds this event and concludes that

the worst case scenario described above results in no significant safety consequences.

3. Leakage past RV-201 is a precursor to a LOCA. The estimated steady state, post trip leakage rate from RV-201 was within the capacity of one charging pump. The effects of this leakage rate are fully bounded by the UFSAR Chapter 14, Safety Analyses concerning LOCAs. Equipment relied on in the Chapter 14 LOCA analyses was available but was generally not challenged by the relatively small leakage of this event.

This event is considered reportable in accordance with 10 CFR 50.73(a)(2)(iv) as an event that resulted in the manual or automatic actuation of any engineered safety feature including the RPS.

IV. CORRECTIVE ACTIONS

Corrective actions fell into two categories: those to facilitate a safe Unit restart (short-term actions), and those to prevent recurrence (long-term actions). A summary of the recommended corrective actions are listed below.

A. TURBINE CONTROL SYSTEM

1. Short-Term Actions

- Replaced the failed SADI board.
- Performed component level checks on the other SADIs in Turbine EHC Cabinet 1T11.
- Replaced failed fans in EHC Cabinet 1T11.
- Repaired all minor deficiencies discovered during troubleshooting activities.
- Sampled the turbine EHC fluid to ensure it was clean.
- Performed a non-destructive failure analysis of the SADI board and servo valve for MTSV-2 to determine any obvious failure mechanisms prior to sending the parts to vendor for a more complete analysis.

2. Long-Term Actions

- Started evaluating methods to periodically check fan operation.
- Sent the SADI board to the vendor for a failure analysis. The failure analysis conducted by the vendor produced no conclusive results. The SADI board was subsequently sent to an independent consultant specializing in the identification of hardware failure modes for more rigorous testing to discover why the component

failed. The results of the root cause analysis are detailed in Section II.

3. The root cause of this event has been incorporated into an ongoing analysis of our turbine EHC systems that was initiated to identify improvements in their operation, maintenance, and design. The recommendations and findings of this root cause analysis will be implemented as appropriate.

4. We plan to replace all transistors on the SADI boards with modern components or replace the SADI boards themselves. The replacement options are currently under evaluation by our Design Engineering Section.

B. REACTOR COOLANT SYSTEM RELIEF VALVES

1. Short-Term Actions

- Replaced RV-201.
- Performed piping and hanger inspections and made appropriate adjustments to the relief valve piping support system.
- Performed a non-intrusive examination on the old RV-201 to determine if any obvious deficiencies exist. None were noted.
- Repaired the leaking ERV-404 pilot valve.
- Performed lift test of RV-200. This was done because the cause of the RV-201 leakage had not been determined and RV-200 was subjected to the same pressure/thermal transient as RV-201.

2. Long-Term Actions

- The valve manufacturer has initiated improvements to the disc holder/bellows assembly staking process to ensure that appropriate quality assurance methods are in place to prevent recurrence of this problem.

V. ADDITIONAL INFORMATION

A. Identification of Components Referred to in this LER.

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Steam Generator	HX	SJ
Auxiliary Feedwater Pump	P	BA
Main Turbine Stop Valve	SHV	TA
Main Turbine EHC System	NA	TG
SADI Board	CBD	TG

B. Previous Similar Events.

There has been one previous similar event at Calvert Cliffs involving the simultaneous closure of all the MTSVs. Details may be found in LER 317/94-006. The cause of MTSV closure in that event was concluded to be the same failure mechanism observed in this event, but no connection between the events has yet been confirmed.

A previous event (LER 317/87-006) reported safety valve leakage where RV-200 leaked under steady state plant conditions. Corrective action for that event included, in part, the replacement of both safety valves.

PRESSURIZER SAFETY VALVE
CALVERT CLIFFS NUCLEAR POWER PLANT

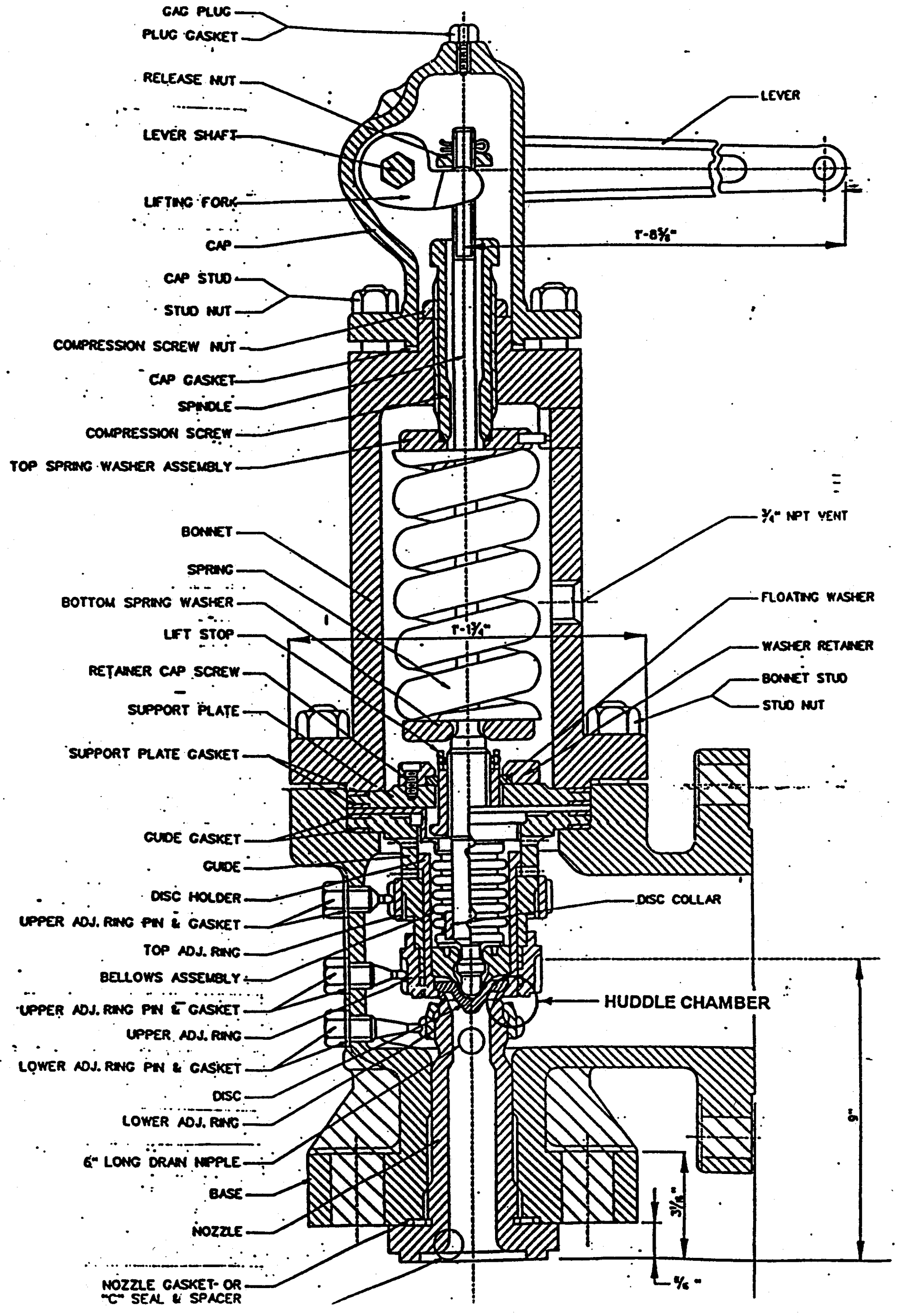

FIGURE 1

NRC FORM 366A (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) **TEXT CONTINUATION**		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST; 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

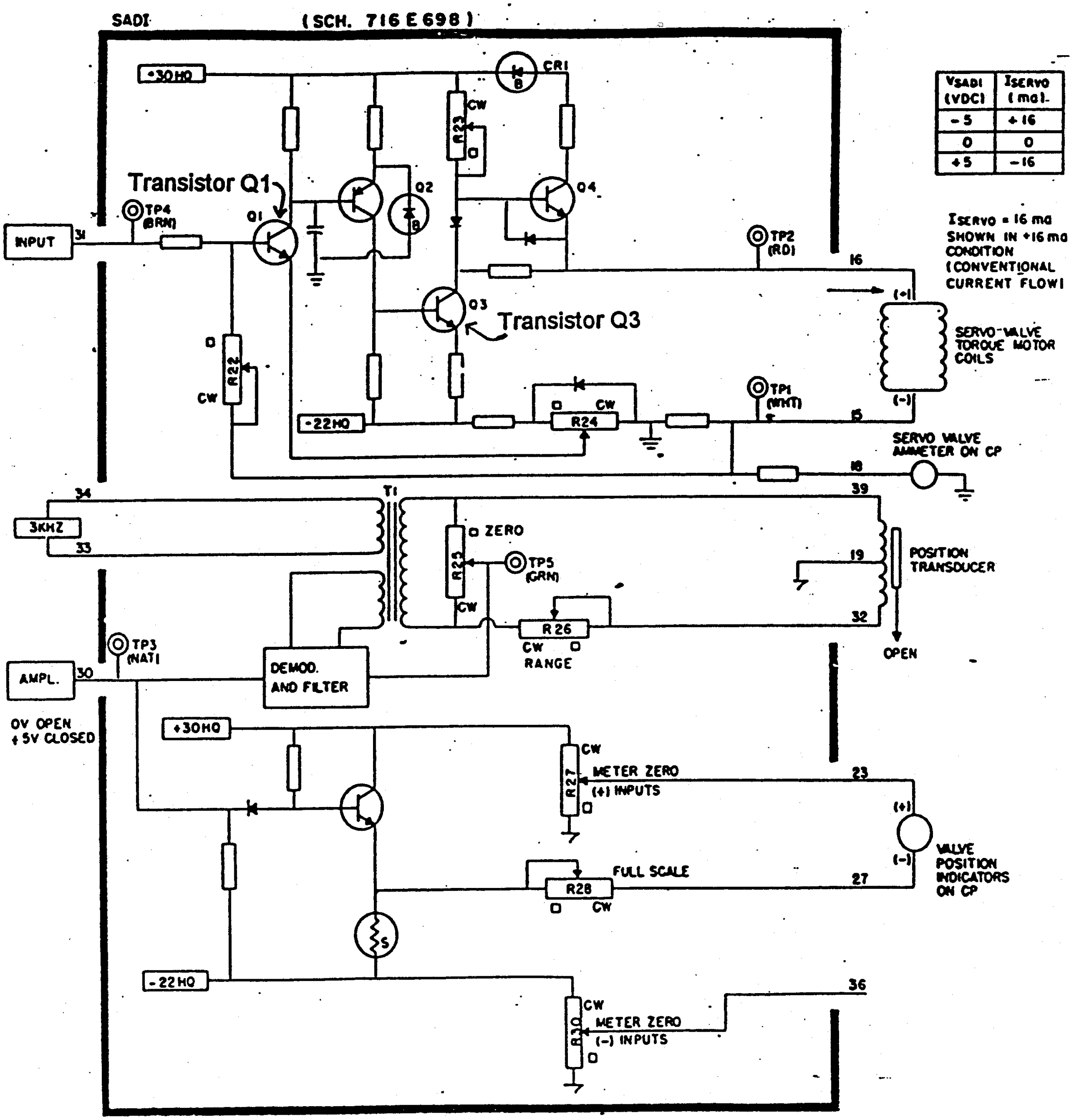

FIGURE 2